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FAST NEUTRON FLUXES IN HANFORD REACTORS

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FAST NEUTRON FLUXES IN HANFORD REACTORS

By

H.H. Yoshikawa

May 3, 1960

Hanford Laboratories Operation
Richland, Washington

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ABSTRACT

The results of a multi-group, diffusion type code computation for the fast neutron flux distribution in C, K, and N reactors are presented. Relative Ni activations in conjunction with the calculated neutron spectra are used to obtain values for the integrated fast flux per MWD/AT for C and K reactors.

The relative, theoretical, integrated fast flux values at the same adjacent fuel powers in C, K, and N reactors in the mid-plane of the filler layers are:

C 1.0
K 1.3
N 2.6

For C reactor, the best estimate of the integrated fast flux per MWD/AT is:

$$5.0 \times 10^{16} \text{ nvt (> 1 Mev) } = 1 \text{ MWD/AT}$$

$$\frac{\text{neutrons}}{\text{cm}^2} \times \frac{\text{cm}}{\text{sec}} = \text{sec}$$

$$\frac{\text{neutrons}}{\text{cm}^2} \times \frac{\text{cm}}{\text{sec}} = \text{sec}$$

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FAST NEUTRON FLUXES IN HANFORD REACTORSI. Introduction

Irradiation effect studies have been performed in various Hanford irradiation test facilities for some time, but exposures at these facilities have been reported in terms of the unit MWD/AT which is derived from the total energy produced in the process tubes adjacent to the experimental facilities. In general, thermal neutron exposures can be accurately estimated from the exposure in MWD/AT units,⁽¹⁾ since the thermal component is well described by reactor theory. However, for most irradiation effect studies in graphite and in metals, the fast neutron flux in the region above 0.1 Mev is generally conceded to be the most important. As will be seen, the relation between MWD/AT and the integrated fast flux is not a constant for all facilities in all reactors, but rather is different for each type of facility in each reactor and perhaps even depends on the position within the facility. Hence, a measure of the number of fast neutrons per MWD/AT in the Hanford irradiation facilities is necessary to obtain a more accurate knowledge of the exposure relevant to radiation effects studies in graphite.

The results of tests at off-site, high intensity, irradiation test reactors are obtained in terms of property changes per unit of energy generation in the test reactor or per unit of foil activation. Thus, in order to apply the results of such tests to Hanford reactors, it is necessary to devise a common unit by means of which rates of damage can be computed. If a value for the fast neutron exposure in the Hanford reactors per MWD/AT is obtained, the results from irradiations at high intensity, test reactors can then be used to calculate rate data applicable to Hanford conditions.

Further, the design of the NPR differs significantly from the design of the existing Hanford reactors. In particular, the uranium to graphite ratio in the NPR is significantly higher, by a factor of two, than in any of the present Hanford reactors. Therefore, it is expected that the number of fast neutrons per MWD/AT_N (where the subscript indicates the reactor for which the unit is used) is larger than in the existing Hanford reactors. In order to correlate expected damage rates of the NPR with existing reactors, it is necessary to know the relative number of fast neutrons per MWD/AT in the NPR and in the existing reactors.

Thus, to determine damaging exposures more accurately, to apply test irradiation data to Hanford reactors and to predict damage rates in the NPR, it is necessary to determine as accurately as possible, if feasible on an absolute basis or if not, on a relative basis, the fast fluxes in the existing Hanford reactors.

The determination of fluxes has proceeded by two methods. In the first method to be treated in Section II, a completely theoretical calculation, based on the lattice dimensions and other physical characteristics of the reactor, yields estimates of the fast flux which are tabulated in Section III. In the second method, found in Section IV, the results of foil activation are used in conjunction with the theoretically derived energy distribution of the neutron flux to obtain an experimentally based estimate of the fast flux.

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II. Theory

Computations were made of the spatial and energy distribution of the fast neutron flux in C, K and N Reactors, using the GNU-II code. This is a 32-group, diffusion-type, one-dimensional code, specifically modified to include twenty groups in the range 10 Mev to .1832 Mev in intervals of 0.2 lethargy units. Such a modification was performed in order to determine more accurately the portion of the neutron flux most important to radiation effect studies. In addition, segregation into a number of energy groups allowed the energy dependence of the total scattering cross section and of the average energy transfer per collision to be included in the calculation.

An infinite cylinder approximation is used for the reactor lattice unit, with the cross sectional area of the cylinder equal to the area of the reactor lattice unit. To approximate the environment in the core of the reactor, the boundary condition at the periphery of the cylinder is that of no net current of neutrons across the outer boundary. Each of the several annular regions within the outer cylinder was characterized by the proper material as indicated in the reactor construction. In general, the fuel element, water, process tube and graphite were incorporated into the cylinder model. The presence of water complicates any diffusion type analysis because of the large possible energy loss by a neutron upon a single collision with hydrogen. However, this code does make a special provision for the presence of hydrogen.

All computations were made on the basis of one fission neutron per cubic centimeter at fuel per centimeter of process tube length. Thus, all output values can be converted to fluxes if the power generation per centimeter of process tube length is known. More conveniently, the integrated neutron flux per unit of energy generation, MWD/AT, can be computed.

As input data, it is necessary to include the lattice size, fuel element configuration and process tube configuration. C Reactor was chosen for one of the computations since the Y test hole has been used extensively for radiation effects studies on graphite at high temperatures. In Figure 1 is shown the arrangement of the reactor lattice and in Figure 10 the equivalent cylindrical approximation employed in the computations. A four-region approximation was used in the calculations with the uranium in the first region, a homogenized mixture of aluminum and water in the second, air in the third, and graphite in the outer annulus. As seen in the figure, a solid fuel element was employed. A tabulation of the input information for all reactors considered in this study is found in Appendix I.

Since the test holes, 2C in KE and 2C in KW, have been activated for use in graphite irradiations, computations were performed for the K Reactor lattice. In Figure 2 is shown the arrangement of the reactor lattice and in Figure 11 the equivalent cylindrical approximation employed in the computation. An I&E fuel element was included, which led to a more detailed cylindrical approximation than for the C Reactor.

Because of the relatively open structure of the NPR moderator stack and because of the rectangular lattice arrangement, two separate computations have been made for the NPR. Figure 3 shows the arrangement of the reactor lattice and Figure 12 the equivalent cylindrical approximation employed in one calculation. As seen in the figure, the cylindrical approximation was quite detailed with twelve regions incorporated into the model. For this computation, a tube-in-tube fuel element was employed and the enrichment was .947 percent.

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In addition to the cylindrical approximation, a two-dimensional calculation of the flux distribution in the rectangular lattice unit was performed using the PDQ code.⁽²⁾ The model for these calculations is found in Figure 13. This PDQ calculation included four-energy groups with the lower energy limit of the first group .1832 Mev. Hence, this first group included the energy region covered by the first twenty groups in the cylindrical approximation. Also, as seen in Figure 13, the model was far less detailed with only four regions, which included a homogenized fuel, zirconium and water region, a homogenized zirconium-water region, a graphite region, and finally a region for the voids in the NPR lattice.

III. Theoretical Results

The results of the computation are the flux values at the various radial positions of the cylinder for each of the energy groups. These data yield both the spectrum at each of the radial positions and the radial dependence of the flux at any energy value.

Figures 4, 5, and 6, which include the fast and epithermal components, indicate that the $1/E$ approximation is not valid for the NPR (i.e., the flux per unit lethargy is not constant), but is quite valid for C and K Reactors. However, from the value of the flux at 120 ev, the energy of the principal resonance absorption in Cobalt-59, it is apparent that a severe underestimation of the epithermal flux component would result from the use of a Cobalt-Cadmium ratio to estimate epithermal fluxes in all three reactors. Figures 7, 8, and 9 are energy distribution plots of the high energy component, $> .1832$ Mev, which is the component of greatest interest for radiation effects studies. It is apparent that the energy distribution of the neutron flux is quite different for these three cases. As expected, since the NPR is under-moderated, the flux spectrum in that reactor is skewed toward higher energies. The relative integrated flux > 1 Mev/MWD/AT for C, K, and N Reactors obtained solely from the computations are shown in Table I.

Although the fast neutron flux in the region above 0.1 Mev is the most important for radiation effect studies, it has become customary to report only integrated fast fluxes above 1 Mev. This unit, nvt > 1 Mev, is used since the results from foil activation at the test reactors are reported in terms of that unit. At these test reactors, the spectrum is believed to be similar to the unmoderated fission spectrum down to about 1 Mev, at which point the epithermal component begins to be appreciable. Since no means have been available for estimating this epithermal component, results from the nickel activation, which actually is sensitive to the component above 4 Mev, have been extrapolated only to 1 Mev. For the Hanford Reactors, the computed spectrum would allow an extrapolation to lower energies if necessary. However, the lower energy limit would not affect the accuracy of exposure estimates if the relative amount of damage caused by neutrons with their energy below 1 Mev is the same for dissimilar spectra, or if the amount of damage caused by neutrons below 1 Mev were negligible. It is expected that the validity of this procedure will depend on the particular irradiation effect being investigated.

TABLE I

THEORETICAL VALUES, NVT > 1 MEV PER MWD/AT

<u>Reactor</u>	<u>nvt > 1 Mev/(MWD/AT)</u>	<u>Relative Value</u>
C	4.7×10^{16}	1.0
K	6.1×10^{16}	1.3
N	12×10^{16}	2.6

2.555

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The radial distribution of the flux at 1 Mev for C, K and N Reactors is shown in Figures 10, 11, and 12. For these three figures, the numerical values for the integrated flux are all based on a common unit. Thus, the relative integrated flux values can be inter-compared among the three reactors. The spatial distribution of the relative magnitude of the integrated flux per MWD/AT energy generation is illustrated in each figure. The flux intensity at the inner-wall of the graphite tube block is 2.0 times the flux at the outer boundary for C compared to values of the same ratio of 1.75 for K and 1.35 for N. The result of the PDQ two-dimensional calculation is shown in Figure 13 for the flux dependence along the vertical line joining two process tubes in the NPR. Since the ratio of the flux at the inner-wall of the tube block to the flux at the tube boundary for the PDQ is 1.4 compared to 1.35 for the one-dimensional calculation, the calculated radial variation is very similar in the two cases. The result of the PDQ calculation for the flux dependence along the horizontal line joining two process tubes is shown in Figure 14.

IV. Experimental

The neutron flux values as determined by foil activation techniques afforded an excellent means of checking the accuracy of the theoretical values. If both the theoretical and experimental numbers were in agreement in the existing reactors, greater confidence could be attached to the theoretical flux estimates in the NPR.

The procedure employed for foil activation studies has been outlined in HW-64251⁽⁴⁾ and consists briefly of irradiating nickel foil monitors in the test facilities and computing the fast neutron exposure from the activity of the foils after discharge. Counting and calibration procedures for determining the activity of the foils are described in HW-64506.⁽⁵⁾ Such activation studies have been performed in a variety of Hanford irradiation facilities and the relative nickel activation per MWD/AT exposure is presented in Table II. Also included are values of the integrated thermal flux obtained by cobalt activation and values of the cobalt-cadmium ratio. All nickel and cobalt activations were corrected for buildup during the irradiation period and decay after discharge until counting, but no burnup corrections were applied because of the small exposure to the foils. Flux depression and self-shielding are negligible because of the small mass, approximately 10 milligrams, of the nickel foil and the small quantity of cobalt, 0.5 w/o, in the aluminum-cobalt monitors, which were about 20 milligrams in weight.

TABLE II

FOIL DATA

<u>Facility</u>	<u>Type</u>	<u>Relative Ni Activation (MWD/AT)</u>	<u>Th. Neutron nvt (MWD/AT)</u>	<u>Cobalt-Cadmium Ratio</u>
Y at C	Hot test hole	1.00	---	---
2C at KE	Hot test hole	1.32	1.7×10^{17}	---
2B at KE	Cooled Test Hole	1.04	1.1×10^{17}	---
1573 DR	Annulus Tube	.54	1.45×10^{17}	13
Snout II	Cooled Test Facility	1.25	1.7×10^{17}	20

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V. Derived Integrated Fluxes

From the foil activations, the integrated fluxes can be obtained if the cross section for the nickel activation is known as a function of energy and the neutron flux energy distribution is known. The neutron flux energy distribution has been obtained from the calculations previously mentioned and the activation cross section for nickel, averaged over an unmoderated fission spectrum, has been determined.⁽³⁾ However, because of the lack of experimental data, it was necessary to use a theoretically derived relation for the energy dependence of the nickel activation cross section, $\sigma_a(E)$.

The form of $\sigma_a(E)$ was obtained by computing the threshold energy for the $\text{Ni}^{58}(n,p)\text{Co}^{58}$ process from the nuclear masses involved. Although the total process is exothermic, even for zero kinetic energy of the incident neutron, the Coulomb barrier imposes a threshold of 6 Mev for the process. Using the proton emission probability function, the decrease in cross section below the threshold energy was computed and the cross section was assumed constant above 6 Mev. The final form of $\sigma_a(E)$ (the energy dependence of the activation cross section) is shown in Figure 15. The details of the computations of $\sigma_a(E)$ are found in Appendix II. The absolute value for $\sigma_a(E)$ was calculated by integrating $\sigma_a(E)$ over the Watt fission spectrum⁽⁶⁾ and normalizing the result to conform to the measured average cross section of 91 millibarns.

An effective threshold energy for the activation can be computed in the manner of Hughes.⁽⁷⁾ This effective threshold energy is defined as follows:

$$\frac{1}{\sigma_a(6 \text{ Mev})} \int_0^{\infty} \sigma_a(E) \phi(E) dE = \int_{E_{\text{eff}}}^{\infty} \phi(E) dE$$

In a sense, this threshold energy defines the lower energy limit of neutrons which are effective in causing activation. For the Watt spectrum, E_{eff} is 4.0 Mev and for the calculated spectrum in C Reactor, the value of E_{eff} is 3.8 Mev. Hence, the threshold energy is fairly independent of the energy distribution of the fast neutrons. The absolute value for $\sigma_a(E)$ was calculated from the observed average cross section of 91 mb⁽⁵⁾ and the Watt fission spectrum.

Using this form for $\sigma_a(E)$, the integral

$$\int \sigma_a(E) \phi(E) dE$$

where $\phi(E)$ is the calculated fast neutron flux, is evaluated graphically to obtain the relationship between the activation in the foil and the total fast integrated flux. To insure consistency in reporting of the flux, only the flux above 1 Mev has been evaluated. For the reactors in which the flux computation have been carried out, it may be feasible to extrapolate the values to lower energies.

The final results for the integrated flux per MWD/AT in the existing Hanford reactors are based on the observed Ni activation modified by both the calculated neutron flux spectrum and the calculated energy dependence of the activation cross section for Ni^{58} . Table III presents these numbers for C and K Reactors. As can be seen, the values for the fast flux in C and K Reactors obtained in the manner described above are about 10 percent higher than the value obtained from the calculations alone. This relatively close agreement increases confidence in the relative magnitude of the integrated fluxes predicted for the NPR compared to C Reactor.

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TABLE III

BEST ESTIMATES FAST NEUTRON EXPOSURE/(MWD/AT)

<u>Test Facility</u>	<u>nvt > 1 Mev/MWD/AT</u>
Y at C	5.0×10^{16}
2C at KE	6.9×10^{16}

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The author is deeply indebted to Dr. C. A. Preskitt of ORNL who performed the computations cited in this report. At HAP0 the assistance of Mr. Lee Duncan of Radiochemical Analysis, who performed the foil counting, and of Mr. A. J. Jervis, who performed many of the foil irradiations, is gratefully acknowledged.

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APPENDIX I

REACTOR SPECIFICATIONS USED IN GNU-II COMPUTATIONS

<u>Moderator Lattice</u>	<u>Reactor</u>		
	<u>C</u>	<u>K</u>	<u>N</u>
Graphite density (g/cm ³)	1.65	1.65	1.65
Lattice Spacing (in)	8-3/8 x 8-3/8	7-1/2 x 7-1/2	8 x 9
Tube Block Cross Section (in)	4-3/16 x 4-3/16	3.690 x 3.630	6 x 6
Filler Block (in)	4-3/16 x 4-3/16	3-3/4 x 3-3/4	6 x 6
<u>Process Channel</u>			
Graphite Bore Diameter (in)	1.778	1.858	3.5
Process Tube (in)	OD 1.764 Al	1.806 Al	3.2 Zr.
	ID 1.640 (avg.)	1.680	2.7
Water Annulus	Cross 0.443 in ²	Cross 0.538 in ²	OD 2.70
	Section	Section	ID 2.42
<u>Fuel Element Dimensions</u>			
<u>(Slug Type)</u>	<u>Solid</u>	<u>I&E</u>	<u>Tube-in-Tube</u>
Avg. Charge U, lbs/ft.	10.5	10.5	25
Enrichment, percent	.72	.72	.947

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APPENDIX II

PROCEDURE FOR CALCULATING $\sigma(E)$

In order to reduce the Ni activation data, the energy dependence for the activation cross section, $\sigma(E)$, must be known. To calculate $\sigma(E)$, it is necessary to compute the incident neutron threshold energy, B_0 , for the reaction $Ni^{58}(n,p)Co^{58}$. B_0 , the threshold energy, can be calculated on the basis of the mass difference of the particles taking part in the reaction. The mass of Co^{58} can be computed, however, by using the mass of Co^{59} , the mass of a neutron and the binding energy of the last neutron in Co^{59} , i.e.,

$$m(Co^{58}) = m(Co^{59}) - m_n + E_b$$

$m(Co^{58})$ is the mass of Co^{58}

$m(Co^{59})$ is the mass of Co^{59}

m_n is the mass of a neutron

E_b is the binding energy of the last neutron in Co^{59}

Values for the masses and binding energy are taken from AIP Handbook⁽⁸⁾

$$\begin{aligned} m_n &= 1.00898 \text{ a.m.u.} \\ E_b &= \frac{.01100}{.99798} (10.25 \text{ Mev}) \end{aligned}$$

$$\begin{aligned} m(Co^{59}) &= 58.95157 \\ &\quad - .99798 \\ m(Co^{58}) &= 57.95359 \end{aligned}$$

Using this mass for Co^{58} , the total mass difference in the reaction $Ni^{58}(n,p)Co^{58}$ can be computed.

$$\begin{array}{r} Ni^{58} \quad 57.95348 \\ + n \quad 1.00898 \\ \hline 58.96247 \end{array}$$

$$\begin{array}{r} Co^{58} \quad 57.95359 \\ + H^1 \quad 1.00815 \\ \hline 58.96174 \end{array}$$

$$\begin{array}{r} 58.96247 \\ - 58.96174 \\ \hline + .00073 \text{ a.m.u.} \end{array}$$

$$= .7 \text{ Mev}$$

Thus, the total reaction is exothermic by the amount 0.7 Mev. However, the proton must surmount a Coulomb barrier before emission. This Coulomb barrier is given by⁽⁷⁾

$$E_c = .96 z Z/A^{1/3} \text{ Mev} \quad z = 1, Z = 27, A = 59$$

$$\begin{array}{r} E_c = 6.70 \text{ Mev.} \quad 6.70 \\ \quad \quad \quad .70 \\ \hline 6.0 \end{array}$$

Thus, one obtains $B_0 = 6.0 \text{ Mev}$ as the threshold energy.

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With this value for B_0 , one can use the formula derived by Bethe⁽⁹⁾ for the proton emission probability for incident neutron energies below 6.0 Mev. This formula is:

$$P = \exp. - \left\{ \frac{2}{\hbar} (2z Z e^2 m_p r)^{1/2} [x^{-1/2} \cos^{-1} x^{1/2} - (1 - x)^{1/2}] \right\}$$

where $x = E/E_c$, $E_c = 6.70$ Mev

$$z = 1$$

$$Z = 27$$

e = electronic charge

m_p = mass of proton

$r = 1.5 \times 10^{-13} \text{ cm}$, nuclear radius

$$\hbar = \frac{h}{2\pi} \quad h \text{ is Planck's constant}$$

This function has been evaluated numerically and the tabulated values are plotted in Figure 15.

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FIGURE 1

C Reactor Lattice and Process Tube Details

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FIGURE 2

K Reactor Lattice and Process Tube Details

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FIGURE 3'

N Reactor Lattice and Process Tube Details

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FIGURE 4

Energy Distribution of the Neutron Flux in C Reactor

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FIGURE 5

Energy Distribution of the Neutron Flux in K Reactor

FIGURE 6

Energy Distribution of the Neutron Flux in N Reactor

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FIGURE 7

Energy Distribution of Fast Neutrons in C Reactor

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FIGURE 8

Energy Distribution of Fast Neutrons in K Reactor

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FIGURE 9

Energy Distribution of Fast Neutrons in N Reactor

FIGURE 10

Radial Distribution of the Flux at 1 Mev in C Reactor

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FIGURE 11

Radial Distribution of the Flux at 1 Mev in K
Reactor

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FIGURE 12

Radial Distribution of the Flux at 1 Mev in N
Reactor

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Spatial Distribution of the Neutron Flux at 1 Mev
in NPR

FIGURE 13
Flux Through a Solid Section of the Reactor Lattice.

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Flux Distribution Through a Section to Void Regions.
FIGURE 14

Spatial Distribution of the Neutron Flux at 1 Mev in N Reactor

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FIGURE 15

Neutron Energy Dependence of the Activation Cross Section for the Ni Threshold Reaction

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